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SOUTHERN COMPANY
Energy to Serve Your World℠

June 27, 2000

LCV-1453

Docket No. 50-424
50-425

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Ladies and Gentlemen:

VOGTLE ELECTRIC GENERATING PLANT
LICENSEE EVENT REPORT 1-00-002
MANUAL REACTOR TRIP
FOLLOWING MAIN STEAM VALVE CLOSURE

In accordance with the requirements of 10 CFR 50.73, Southern Nuclear Operating Company hereby submits a Vogtle Electric Generating Plant licensee event report for an event that occurred on Unit 1 on June 5, 2000.

Sincerely,

J. B. Beasley, Jr.

JBB/JPC

Enclosure: LER 1-00-002

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser
Mr. M. Sheibani
SNC Document Management

U. S. Nuclear Regulatory Commission
Mr. L. A. Reyes, Regional Administrator
Mr. Ramin R. Assa, Vogtle Project Manager, NRR
Mr. J. Zeiler, Senior Resident Inspector, VEGP

IE22

NRC FORM 366 (6-1998)

U.S.NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104 EXPIRES: 06/30/2001

Estimated burden per response to comply with this mandatory information request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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TITLE (4)

MANUAL REACTOR TRIP FOLLOWING MAIN STEAM ISOLATION VALVE CLOSURE

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										0 5 0 0 0
0 6	0 5	2 0 0 0	2 0 0 0	0 0 2	0 0	0 6	2 7	2 0 0 0	FACILITY NAME	0 5 0 0 0

OPERATING MODE (9)	1
POWER LEVEL (10)	1 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

20.2201(b)	20.2203(a)(2)(v)	50.73(a)(2)(i)	50.73(a)(2)(viii)
20.2203(a)(1)	20.2203(a)(3)(i)	50.73(a)(2)(ii)	50.73(a)(2)(x)
20.2203(a)(2)(i)	20.2203(a)(3)(ii)	50.73(a)(2)(iii)	73.71
20.2203(a)(2)(ii)	20.2203(a)(4)	X 50.73(a)(2)(iv)	OTHER
20.2203(a)(2)(iii)	50.36(c)(1)	50.73(a)(2)(v)	*Specify in Abstract below*
20.2203(a)(2)(iv)	50.36(c)(2)	50.73(a)(2)(vii)	*or in NRC Form 366A*

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (include area code)
Mehdi Sheibani, Nuclear Safety and Compliance	7 0 6 - 8 2 6 - 3 2 0 9

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
B	S B	F U	C 6 3 4	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-space typewritten lines) (16)

On June 5, 2000, at 1140 EDT, the main steam isolation valve (MSIV) for steam generator (SG) #4 closed. Control room operators observed the valve's trouble alarm and also observed the water level decreasing in SG #4. The reactor operator manually tripped the reactor and all rods fully inserted. The unit was stabilized in mode 3 (hot standby).

The direct cause of the reactor trip was the closure of the MSIV which was the result of a blown fuse. The investigation into the cause of the blown fuse found no electrical ground in the circuit or evidence of an overcurrent condition. The area around the valve was also checked for signs of water, loose connections or visible damage, with none found. Therefore, the cause of the fuse failure is unknown. The blown fuse and other fuses were replaced and the unit was returned to service. Other actions including component testing and evaluations are planned to determine the cause and prevent recurrence.

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TEXT (If more space is required, use additional copies of NRC Form 366A)(17)

A. REQUIREMENT FOR REPORT

This report is required per 10 CFR 50.73 (a)(2)(iv) because an unplanned actuation of the reactor protection system (RPS) occurred.

B. UNIT STATUS AT TIME OF EVENT

At the time of this event, Unit 1 was in Mode 1 (Power Operation) at 100 percent of rated thermal power. Other than that described herein, there was no inoperable equipment that contributed to the occurrence of this event.

C. DESCRIPTION OF EVENT

On June 5, 2000, at 1140 EDT, main steam isolation valve (MSIV) 1HV-3036A closed. Control room operators observed the valve's trouble alarm and also observed the water level decreasing in steam generator (SG) #4. The reactor operator manually tripped the reactor and all rods fully inserted. The main feedwater system isolated and the auxiliary feedwater system actuated as designed. A non-1E 4160 volt AC electrical bus failed to properly transfer to its alternate source and was de-energized. The unit was stabilized in mode 3 (hot standby). The NRC Operations Center was notified of this event at 1352 EDT.

D. CAUSE OF EVENT

The direct cause of the reactor trip was the closure of the MSIV which was the result of a blown fuse in the positive side of the circuit. The investigation into the cause of the blown fuse found no electrical ground in the circuit or evidence of an overcurrent condition. The area around the valve was also checked for signs of water, loose connections or visible damage, with none found. A review of the valve's work order history found that the negative side fuse in the MSIV circuit had blown and been replaced during testing in the previous refueling outage. But no firm evidence was found that this previous event had degraded the positive side fuse's ability to operate at rated current levels. Therefore, the cause of the fuse failure is unknown. Further troubleshooting and investigation of the root cause of this event is in progress. This includes performing failure and comparative analyses on the failed fuse and similar fuses. This is expected to be completed by July 31, 2000. If a definitive root cause is determined, a revision to this LER will be submitted.

The cause and corrective actions for the non-1E bus' failure to transfer are being addressed by the plant's event review program.

E. ANALYSIS OF EVENT

When the MSIV closed, control room operators acted appropriately to manually trip the reactor and prevent a challenge to the automatic trip actuation circuitry. The main feedwater system isolated and the auxiliary feedwater system actuated as designed. The unit was stabilized in mode 3 (hot standby). Based on these considerations, there was no adverse effect on plant safety or on the health and safety of the public as a result of this event.

F. CORRECTIVE ACTIONS

The blown positive side fuse as well as the negative side fuse was replaced in the MSIV circuit. Additionally, both fuses were replaced in other MSIVs that had blown a fuse in one side of their circuit during testing in the previous refueling outage. Other actions planned to be taken to prevent recurrence include reviewing maintenance practices, fuse criteria and the MSIV circuit design to determine if improvements are needed. These actions are expected to be completed before the Fall 2000 refueling outage. Further corrective actions may be instituted based on the result of the investigation into the root cause of this event.

G. ADDITIONAL INFORMATION

1) Failed Components:
   3-ampere fuse manufactured by Bussman Fuses, a division of Cooper Industries.
   Model number: FNQ-3.

2) Previous Similar Events:
   LER 50-424/1996-006, dated June 17, 1996.
   Failure of a similar fuse led to closure of a main feedwater isolation valve. Corrective actions included instituting the use of a new type of Bussman FNQ-3 fuse. NOTE: This new type of fuse failed in the June 5, 2000, event.

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TEXT (If more space is required, use additional copies of NRC Form 366A)(17)

3) Energy Industry Identification System Code:
Main Steam System - SB
Main Feedwater System – SJ
Auxiliary Feedwater System - BA